July 30, 2001

Mr. Harold W. Keiser
Chief Nuclear Officer & President
PSEG Nuclear LLC - X04
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION, ISSUANCE OF AMENDMENT RE: 1.4% INCREASE IN LICENSED POWER LEVEL (TAC NO. MB0644)

Dear Mr. Keiser:

The Commission has issued the enclosed Amendment No. 131 to Facility Operating License (FOL) No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications (TSs) and FOL in response to your application dated December 1, 2000, as supplemented on February 12, May 7, and May 14, 2001.

This amendment increases the licensed power level by approximately 1.4% from 3,293 megawatts (MW) thermal to 3,339 MW thermal. The changes are anticipated to increase the unit's net electrical output by 15 MW electric. The changes are based on the installation of the CE Nuclear Power LLC Crossflow ultrasonic flow measurement system and its ability to achieve increased accuracy in measuring feedwater flow.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Federal Register.

Sincerely,

/RA/

Richard B. Ennis, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures: 1. Amendment No. 131 to License No. NPF-57
2. Safety Evaluation

cc w/encls: See next page

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SCollins/BSheron	OPA	OGC	ACRS	

ACCESSION NUMBER: ML011910345 * SE input provided - no major changes made. ** See previous concurrence

OFFICE	PDI-2/PM	TECH ED**	PDI-2/LA	SRXB/SC*	SRXB/SC*	SPLB/SC*	SPSB/SC(A)*	EMCB/SC*	EMEB/SC*
NAME	REnnis	PKleene	TClark	FAkstulewicz	RCaruso	GHubbard	MCaruso	KWichman	KManoly
DATE	7/25/01	7/11/01	7/26/01	5/18/01	6/21/01	6/14/01	5/2/01	5/30/01	6/19/01

OFFICE	EEIB/SC*	EEIB/SC*	IOLB/SC*	OGC	PDI-2/SC	PDI/D	DLPM/D	ADPT/AD	NRR/D
NAME	CHolden	EMarinos	DTrimble	RWeisman	JClifford	EAdensam	JZwolinski	ZWolinski for BSheron	SCollins
DATE	2/14/01	5/23/01	5/17/01	7/25/01	7/26/01	7/26/01	7/26/01	7/26/01	7/27/01

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cc:

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PSEG NUCLEAR LLC

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 131
License No. NPF-57

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by the PSEG Nuclear LLC dated December 1, 2000, as supplemented on February 12, May 7, and May 14, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Operating License and the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-57 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 131, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance, and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility
Operating License and
Technical Specifications

Date of Issuance: July 30, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 131

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following pages of the Facility Operating License and Appendix "A" Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License, page 3	License, page 3
xi	xi
1-6	1-6
3/4 4-22	3/4 4-22
3/4 4-23	3/4 4-23
3/4 4-23a	3/4 4-23a
3/4 4-23b	3/4 4-23b
B 3/4 4-5	B 3/4 4-5
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
6-21	6-21
6-26	6-26

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 131 TO FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

ATLANTIC CITY ELECTRIC COMPANY

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated December 1, 2000, as supplemented by letters dated February 12, May 7, and May 14, 2001, PSEG Nuclear LLC (PSEG or the licensee) submitted a request for changes to the Hope Creek Generating Station (HCGS) Facility Operating License (FOL) and Technical Specifications (TSs). This proposed amendment would increase the licensed power level by approximately 1.4% from 3,293 megawatts thermal (MWt) to 3,339 MWt. The proposed changes are based on the installation of the CE Nuclear Power LLC (CENP) Crossflow ultrasonic flow measurement (UFM) system and its ability to achieve increased accuracy in measuring feedwater flow.

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specific core thermal power level. The power level is indicated in the control room by neutron flux instrumentation that is calibrated to correspond to core thermal power. Core thermal power is validated by a nuclear steam supply system (NSSS) energy balance calculation. The accuracy of this calculation depends primarily upon the accuracy of feedwater flow, temperature, and pressure measurements.

The thermal power levels assumed in a plant's design basis transient and accident analyses must bound the potential range of power levels at which the plant could be operated. The uncertainty of calculating values of core thermal power is factored into the allowable thermal power levels to reduce the likelihood of exceeding the power levels assumed in the analyses. At one time, Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix K, required licensees to base their transient and accident analyses on an assumed power level of at least 102% of the licensed thermal power level. This was to allow for uncertainties in determining thermal power (e.g., instrument measurement uncertainties). The 2% power margin uncertainty value was intended to address uncertainties related to heat sources in addition to instrument measurement uncertainties. The U.S. Nuclear Regulatory Commission (NRC) concluded, at the time of the original emergency core cooling system (ECCS) rulemaking, that the 2% power margin requirement was based solely on considerations associated with power measurement uncertainty, as is reflected in Appendix K.

Appendix K to 10 CFR Part 50 did not require demonstration of the power measurement uncertainty and mandated a 2% margin, notwithstanding that the instruments used to calibrate the neutron flux instrumentation may be more accurate than originally assumed in the ECCS rulemaking. On June 1, 2000, the NRC published a final rule (65 FR 34913) that allows licensees to justify a smaller margin for power measurement uncertainty by using more accurate instrumentation to calculate the reactor thermal power and thereby calibrate the neutron flux instrumentation.

In its application, PSEG requested approval to increase the HCGS licensed thermal power level based on the installation of the CENP Crossflow UFM system. The Crossflow system is designed to improve the accuracy of feedwater flow rate measurement, which is used, in part, to calculate reactor thermal power. The improved flow measurement instrumentation would allow PSEG to operate HCGS with a reduced margin between the actual power level and the 102% power level previously used in the licensing basis ECCS analyses.

3.0 EVALUATION

The NRC staff's review of the licensee's application is organized as follows:

3.1 Reactor - Core and Fuel Performance
3.2 Reactor Coolant System and Connected Systems
3.3 Engineered Safety Features
3.4 Instrumentation and Controls
3.5 Electrical Systems
3.6 Auxiliary Systems
3.7 Steam and Power Conversion Systems
3.8 Radioactive Waste Management
3.9 Radiological Consequences
3.10 Human Factors
3.11 Accident Analysis
3.12 Other Evaluations
3.13 FOL and TS Changes
3.14 Evaluation Summary

3.1 Reactor - Core and Fuel Performance

3.1.1 Reactor - Core and Fuel Performance - Background

The core thermal-hydraulic design and fuel performance characteristics are evaluated for every reload in accordance with the methodology described in CENPDS-839-P, Revision 0, "Westinghouse BWR Reload Licensing Methodology Basis for Public Service Electric & Gas Hope Creek Generating Station" dated August 2000. The slightly higher thermal power requested by the licensee could affect the core and fuel performance, so the licensee evaluated whether the thermal power increase affected the validity of the thermal-hydraulic design models, the critical power ratio (CPR) correlations, the safety limit minimum critical power ratio (SLMCPR) evaluation, and the thermal-hydraulic design evaluations. The following sections address the impact of the power uprate on the fuel design performance, thermal limits, power/flow map, and reactor stability.

3.1.2 Fuel Design and Operation

The licensee stated that the mechanical design and licensing criteria for the current HCGS core (consisting of GE 9B and SVEA-96+ fuel rods) are satisfied according to the NRC-approved methodology,

(1) if the fuel rod linear heat generation rates (LHGRs) are maintained below the steady state LHGR limits (The operating limit LHGR is a core operating limit that ensures the fuel thermal-mechanical performance limit (i.e., the 1% fuel plastic strain design limit or the no fuel centerline melt limit) will not be exceeded as a result of an anticipated operational occurrence (AOO)), and

(2) if the transient fuel rod LHGR overpowers are no more than a certain percent overpower above the steady state limits, and

(3) if the fuel rod burnups do not exceed the existing NRC-approved design or application burnup limitations.

The 1.4% power uprate does not require new fuel designs. The licensee stated that the fuel-specific steady state operating limit LHGR, the transient overpower limit, and the fuel burnup limit will not be changed by the power uprate. Since these limits will not be changed, the staff concludes that the existing fuel design and operation for the SVEA-96+ and GE9B fuel will continue to satisfy the mechanical design and licensing criteria in accordance with an NRC-approved methodology and NRC-approved computer codes at the uprated power conditions.

3.1.3 Thermal Limits Assessment

General Design Criterion (GDC) 10 of 10 CFR Part 50, Appendix A, requires, in part, that the reactor core and associated control and protection systems be designed with appropriate margin to ensure that the specified acceptable fuel design limits are not exceeded during normal operation, including the effects of AOOs. Operating limits are established to ensure that regulatory and/or safety limits are not exceeded for a range of postulated events (transients and accidents). The SLMCPR is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated. The operating limit minimum critical power ratio (OLMCPR) is established to ensure that no fuel damage results during AOOs.

The licensee stated that the SLMCPR is analyzed for each reload core and that the power uprate condition does not generically affect the SLMCPR methodology or bases. PSEG evaluated the validity of the CPR correlation for the SVEA-96+ fuel and the GE9B fuel under the uprated conditions and determined that the applicability range for both correlations (ABBD2.0 for SVEA-96+ fuel and US96G9 for the GE9B fuel) bounds the operating range for both normal operation and AOO conditions.

The steady state operating limit LHGR for each fuel type and the fuel burnup limit will not be changed by the power uprate. Therefore, the uprate has no generic impact on the LHGR acceptance criteria. For every reload, the licensee will determine the change in the LHGR for the limiting transients and ensure that the calculated LHGR is less than or equal to the LHGR overpower limit. Thus, the licensee will establish the OLMCPR, the SLMCPR, and the operating LHGR for each reload using NRC-approved methodology and the steady state thermal-hydraulic core conditions will be analyzed at the maximum power that bounds the uprated power.

The maximum average planar linear heat generation rate (MAPLHGR) operating limit is based on the most limiting LOCA and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46. For every new fuel type, the licensee performs LOCA analyses to confirm compliance with the LOCA acceptance criteria, and for every reload the licensee also confirms that the MAPLHGR operating limit for each reload fuel bundle design remains applicable. The LOCA analyses for HCGS are based on 104.2% of the current rated thermal power level and 105% of the current design core flow, which bound the maximum operating condition for the proposed power uprate.

Based on the above, the staff concludes that the proposed power uprate is acceptable with respect to the fuel design limits.

3.1.4 Power/Flow Operating Map

The licensee stated that the power uprate will not increase the licensed maximum core flow or the operating domain of the power/flow map, but the associated control and protective systems, which are based on percent power and percent flow, will be rescaled to the uprated thermal power. The licensee also stated that the steady state thermal-hydraulic analysis for HCGS covered the operating domain in the power/flow map up to 108% power and 104% flow where 100% power corresponds to 3,293 MWt and 100% flow corresponded to 100 million pounds-mass/hour. Therefore, the 1.4% power uprate will remain within the range of the current steady state core thermal-hydraulic analysis.

HCGS is licensed to operate with an increased core flow of 105% and an extended load line limit analysis (ELLLA) region. The proposed power uprate will extend the ELLLA region to the 100% uprated power level. Thus, the power/flow map will have a smaller range of core flows at the uprated 100% power level.

Based on the above, the staff concludes that the proposed power uprate is acceptable with respect to the power/flow operating map.

3.1.5 Stability

In response to Generic Letter 94-02, PSEG installed an oscillation power range monitor (OPRM) to automatically detect and suppress instability, in accordance with Option III of the BWR Owners Group (BWROG) recommendations (NEDO-31960). The licensee is implementing the BWROG interim corrective actions (ICAs) until the system is tested. The ICAs, in part, identify a restricted and an exclusion region on the power/flow map and the ICAs also delineate operator actions to scram, avoid, or exit the stability regions. The licensee performs analyses that confirm the stability boundary defined in HCGS TS Figure 3.4.1.1-1 at several power/flow state-points and burnups throughout the cycle. The licensee does not apply an explicit power measurement uncertainty to any of the existing stability boundary analyses, since the objective of the analyses is to confirm the existing stability boundary. PSEG credits conservatism in the stability analysis parameters and methods, stating that there is enough conservatism in the analysis methodology described in CENPD-295-P-A to eliminate the need for power measurement uncertainty. PSEG concluded that, based on conservatism, the Westinghouse BWR stability methodology is unaffected by the proposed power uprate and that the current analysis remains bounding. The licensee added that the instability boundary or inputs to the OPRM will continue to be confirmed for each cycle.

The staff is currently reviewing a November 29, 2000, PSEG amendment request (NRC TAC No. MB0589) to implement the OPRM automatic features for detecting or suppressing power or thermal-hydraulic instability. Currently, PSEG relies on operator action to implement the OPRM system as part of the interim corrective actions. The staff will review the OPRM automatic features and the adequacy of PSEG's stability plan and analytical methods as part of the review of the November 29, 2000, amendment request.

For the proposed power uprate, the extended load limit line will not be changed and the licensee will continue to evaluate any impact the cycle operating domain or core reload will have on the instability boundaries at several power/flow state-points and burnups throughout the cycle. Therefore, the staff concludes that the 1.4% power uprate will not significantly affect the HCGS instability detection and suppression capability.

3.1.6 Reactor - Core and Fuel Performance - Conclusion

Based on the evaluation in Sections 3.1.1 through 3.1.5 of this safety evaluation, the staff concludes that the proposed power uprate is acceptable with respect to its impact on the reactor core and fuel performance.

3.2 Reactor Coolant System and Connected Systems

3.2.1 Nuclear System Pressure Relief

The safety/relief valves (SRVs) provide overpressure protection for the NSSS during abnormal operational transients. The licensee stated that the proposed power uprate will not require an increase in the operating dome pressure. Instead, the licensee will open the turbine control valves position more, allowing more steam to be withdrawn for power generation. Consequently, the SRV setpoints will not be changed. The licensee also stated that adequate margin is available to avoid inadvertent SRV actuation at the current SRV setpoints.

Since the SRVs will actuate at the current setpoints and the current American Society of Mechanical Engineers (ASME) overpressure protection analyses are based on 102% of the current rated power level, the staff concludes that the SRVs will have sufficient capacity to handle the 1.8% increase in the steam flow associated with the proposed uprate.

3.2.2 Reactor Overpressure Protection

The ASME Code allowable peak pressure for the reactor vessel is 1375 psig (110% of the design pressure of 1250 psig), which is the acceptance limit for pressurization events. The licensee analyzes AOOs that may result in the largest overpressure transient on a cycle-specific basis. The most limiting overpressure transient event for HCGS is a main steam isolation valve (MSIV) closure coupled with the failure to scram on MSIV position. The MSIV closure event was analyzed at 102% of the current rated power level to account for power uncertainty. The licensee reported that the corresponding peak vessel bottom pressure for the MSIV closure event was 1331 psig, which remains below the ASME Code allowable peak pressure of 1375 psig. The licensee stated that since the nominal SRV setpoints and reactor operating parameters will not change, the power uprate does not affect the reactor overpressure protection analysis.

The licensee also proposed to rescale the average power range monitor (APRM) high neutron flux setpoint to 118% of the uprated thermal power, which would delay the high neutron flux scram. The licensee stated that the heat transfer correlations used in the current analysis of record will remain applicable at uprated conditions and that the limiting overpressure transient analysis is based on 102% of the current power. The licensee added that the analysis of record shows that the neutron flux increases rapidly (211% rated at 2 seconds) and the maximum vessel pressure occurs at 3 seconds. The licensee concluded that the current overpressure analysis will remain applicable and that raising the high neutron flux APRM scram setpoint will not result in a more severe overpressure because the rapid rate at which neutron flux increases is offset by the slower thermal response of the addition of energy to the coolant.

Raising the high neutron flux scram setpoint will affect the pressurization events, including MSIV closure and generator load rejection without bypass. The MSIV closure event analysis credits the high neutron flux scram and the SRVs for depressurization. The licensee pointed out that in the current analysis of record, there is a 1-second delay between the high neutron flux scram and the peak pressure; however, the analysis is based on a lower scram setpoint. The higher APRM high-flux setpoint will delay scram and may yield higher peak pressures. In the May 7, 2001, supplement the licensee stated that reload core analyses are performed to assure that the proposed reload design and the TS limits are below the ASME Code limits. The licensee will analyze the limiting MSIV closure event with the rescaled APRM flux scram setpoint before implementing the power uprate.

Based on the licensee’s Cycle 10 overpressure analysis, the staff agrees that (1) the SRVs have sufficient capacity to depressurize the reactor vessel at the uprate conditions, (2) there is sufficient margin in the current limiting overpressure transient analysis to account for the power uprated conditions, (3) maintaining the current SRV setpoints will provide prompt depressurization to offset the slight increase in steam generation under the uprated conditions, and (4) the overpressure protection analysis is based on 102% of the current rated thermal power, which bounds the uprated conditions.

3.2.3 Reactor Vessel and Internals

The licensee evaluated the reactor vessel and internal components considering the changes of the design input parameters and loads due to the proposed 1.4% power uprate. The loads applicable to the internal components include reactor internal pressure difference, LOCA, SRV, seismic, annulus pressurization, jet reaction, and fuel lift loads.

The licensee stated that the proposed power uprate does not change the operating reactor pressure and temperature from the current operating condition. The uprate will not change the fuel lift and seismic loads. The LOCA and SRV loads were analyzed at 102% of the current rated power level, which bounds the proposed power uprate. Also, the recirculation design flow does not change for the power uprate. The most limiting annulus pressurization and jet reaction are not affected by the proposed power uprate. The licensee also stated that the reactor internal pressure differences for the normal operating, upset, emergency, and faulted conditions are not affected by the proposed 1.4% power uprate. Therefore, the licensee concluded that the design basis stresses and fatigue usage factors for the reactor vessel and internal components will remain unchanged for the proposed 1.4% power uprate. For the same reasons, the staff agrees with the licensee's conclusion.

The licensee assessed flow-induced vibration for the proposed power uprate. The licensee concluded that the flow-induced vibration loads will remain within the design limits since the maximum core flow and the maximum recirculation drive flow will remain unchanged following the proposed 1.4% power uprate. For the same reasons, the staff agrees with the licensee's conclusion.

3.2.4 Reactor Vessel Fracture Toughness

Appendix G to 10 CFR Part 50 specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected to over its service lifetime. Appendix G requires that certain pressure-temperature (P-T) limits for reactor pressure vessels must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code.

The licensee's application included revised P-T limit curves (TS Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3) that were developed based on the fracture toughness requirements in 10 CFR Part 50, Appendix G; ASME Code Section XI, Appendix G; and ASME Code Cases N-588 and N-640. The use of the code cases was approved by an exemption to NRC regulations dated July 12, 2001. The proposed P-T limit curves are evaluated in Section 3.13 of this safety evaluation.

3.2.5 Reactor Recirculation System

The power uprate will be accomplished by operating along extensions of rod and core flow lines on the power/flow map. HCGS is currently licensed to operate up to a maximum core flow of 105% of the rated flow. The power uprate does not increase the maximum allowable core flow. Therefore, the reactor recirculation flow will be maintained according to the existing power/flow map, with 100% power corresponding to the uprated power level.

The licensee stated that each recirculation pump motor is a vertical, variable speed, alternating current (AC) motor that can drive the pump over a controlled range of 20% to 115% of rated pump speed. Therefore, no change in the recirculation pump flow design bases is required for the variations in the core thermal-hydraulic conditions (higher voids and two-phase pressure drops) due to the power uprate.

Based on our review of the licensee's application, as set forth above, the staff concludes that the proposed power uprate is acceptable with respect to the design and operation of the reactor recirculation system.

3.2.6 Reactor Coolant Piping and Balance of Plant (BOP) Piping

The licensee evaluated the effects of the proposed 1.4% power uprate conditions on the reactor coolant piping and components with regard to higher flow rate, temperature, and pressure. The effects include thermal expansion, dynamic loads, and fluid transient loads on the Class 1 reactor coolant pressure boundary piping systems. The in-line components include equipment nozzles, valve and flange connections, and pipe supports. The licensee indicated that there are no changes in the reactor coolant system operating and design pressures and temperatures, nor are there any changes in the main steam operating and design pressures and temperatures. There is a slight increase in the feedwater system operating pressure and temperature but no change in the design pressure (1,500 psig) and temperature (425 °F). The licensee reviewed the design basis calculation of the NSSS piping and its support components. The licensee found that there are no changes in the piping design pressures and temperatures, and there are no changes in the seismic and hydrodynamic design loads. The only load that was found to be affected by the proposed 1.4% power uprate is the fluid transient load on steam lines due to main steam stop valve closure. The effect of this fluid transient load was found to be bounded by the design conditions. Therefore, the licensee concluded that the existing design bounds the 1.4% uprate conditions. Accordingly, the staff agrees with the licensee's conclusion that the design of piping, components, and their supports is adequate to maintain the structural and pressure boundary integrity of the reactor coolant loop for the proposed 1.4% power uprate condition.

The licensee evaluated the BOP piping systems by comparing the original design basis conditions with those for the proposed power uprate. The BOP piping systems that are affected were determined from the uprated reactor and BOP heat balances. The systems affected by the proposed power uprate are the main steam, extraction steam, turbine bypass, condensate, and feedwater lines.

Based on its review of the existing design basis calculation, the licensee stated that the BOP systems and their components were originally designed for turbine valves-wide-open (VWO) conditions. The turbine VWO condition equates to about 105% steam flow at about 104.2% of the current rated power level. The proposed 1.4% power uprate increases the steam flow by about 1.8%, which is bounded by the VWO condition. The original piping design parameters (temperature and pressure) also envelop the VWO parameters. Therefore, piping and component design parameters (temperature, pressure, and flow) remain bounded by the original design. The licensee indicated that the piping stress analyses of record were also reviewed. The input parameters (temperature and pressure) for the piping stress analyses, which used the original design values, also remained bounding. No new postulated pipe break locations were identified in any of the systems evaluated. The licensee concluded that the BOP piping and components will continue to maintain their structural integrity for the proposed power uprate condition. For the resons set forth above, the NRC staff agrees with the licensee's conclusion that the BOP systems will operate at the proposed power uprate conditions without adverse effects on the piping system and pipe supports.

The licensee's application stated that the power uprate will result in an increase in flow of approximately 1.8% in systems associated with the turbine cycle (i.e., condensate, feedwater, and main steam). Although operation of these systems will remain bounded by the 105% current design flow, higher flow velocities due to the power uprate may increase flow-accelerated corrosion (FAC) in some components. The anticipated increase in FAC caused by a 1.8% flow change will be very small; however, the licensee committed in its December 1, 2000, letter to update its FAC program to incorporate the increased process flow values for the condensate, feedwater, and main steam systems and their subsystems. The staff finds the licensee's proposed actions with respect to the FAC program to be acceptable.

3.2.7 MSIVs

The MSIVs are engineered safety features for the reactor coolant pressure boundary. Within a TS-defined time interval (usually 3 to 5 seconds), the MSIVs close to isolate the reactor vessel during postulated transient and accident conditions. The licensee stated that the MSIV design pressure (1250 psig), temperature (575 °F), and flow ($3.72x10^6$ lbs/hr) bound the maximum uprated operating condition.

PSEG also stated that the maximum flow rate and maximum differential pressure of the MSIVs depend on the operating dome pressure during a postulated steamline break and on the design of the flow restrictor. The licensee pointed out that the power uprate will not change either the maximum dome pressure, the flow restrictor design, or the setpoints for the high flow differential pressure switches that provide MSIV closure. In the May 7, 2001, supplement, the licensee stated that the MSIVs will experience a higher flow rate; however, since MSIV closure is flow assisted, the increased flow will assist closure. The licensee therefore concluded that the MSIV closure function and time will not be affected by the power uprate.

The staff agrees that the operating changes due to the power uprate will have little effect on the structural integrity or the isolation function of the MSIVs. The uprated operating conditions remain bounded by the MSIV normal design conditions. The 1.8% increase in steam flow may slightly increase the pressure drop across the MSIVs. However, MSIVs are generally designed to close against a much higher pressure differential at a higher steam flow. Moreover, the slight increase in the steam flow rate will not significantly affect the flow restrictors because the flow restrictors are designed to accommodate a pressure differential due to postulated line break.

In addition, various TS surveillances require routine monitoring of the MSIV closure time and leakage to ensure that the original licensing basis for the MSIVs is preserved.

3.2.8 Reactor Core Isolation Cooling (RCIC) System

The RCIC system provides core cooling when the reactor pressure vessel (RPV) is isolated from the main condenser and the RPV pressure is greater than the maximum allowable for starting a low pressure core cooling system. The RCIC system is designed to provide rated flow over a range of reactor pressures from 150 psig to the maximum pressure corresponding to the lowest opening setpoint for the SRVs.

The licensee stated that the transient safety analyses in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR, Table 15.0-3) assume that the RCIC system will operate and provide core cooling water with the reactor initially operating at 104.3% of current rated power. The licensee added that since the maximum operating dome pressure and the SRV opening setpoints will not change for the power uprate, the RCIC system will continue to maintain an adequate water level at the proposed power uprate conditions. With the RCIC analyzed to inject during transient events initiated at 104.3% of current rated power and no change to the operating dome pressure and SRV setpoints, the staff agrees with the licensee that the RCIC injection capability is not affected significantly.

3.2.9 Residual Heat Removal (RHR) System

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary system decay heat removal after reactor shutdown for both normal and post-accident conditions. The RHR system is designed to operate in the low pressure coolant injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The effects of the power uprate on the shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode are discussed below. The LPCI mode of operation is evaluated in Section 3.3.2.

Shutdown Cooling Mode

As discussed in UFSAR Section 5.4.7.1.1.1, the design basis requirement of the shutdown cooling mode is to have the capability to remove decay and sensible heat from the reactor primary system so that the reactor outlet temperature is reduced to 125 °F within 20 hours after the control rods have been inserted. The licensee stated that this design requirement can currently be achieved in less than 7 hours. Since the amount of decay heat increases proportionally to the operating reactor power level, the time required to reduce the reactor temperature to shutdown conditions at a 1.4% uprated power level, would only increase slightly. The staff concludes there is sufficient margin in the shutdown cooling mode design for the system to continue to meet its design basis requirement.

Suppression Pool Cooling Mode

As discussed in UFSAR Section 5.4.7.1.1.3, the design basis requirement for the suppression pool cooling mode is to ensure that the suppression pool temperature does not exceed its maximum temperature limit of 170 °F immediately after a blowdown. The licensee stated that the existing analyses associated with the suppression pool cooling mode were evaluated at 102% of the current rated power level (i.e., are bounding for the 1.4% uprated power condition). Therefore, the staff concludes that the proposed power uprate will have no impact on the capability of the suppression pool cooling mode to meet its design basis requirement.

Containment Spray Cooling Mode

As discussed in UFSAR Section 5.4.7.1.1.4, the design basis requirement for the containment spray cooling mode is that there are two redundant means to spray into the drywell and suppression pool vapor space to reduce internal pressure to below design limits. Since the proposed power uprate does not change the functional design or operation of the RHR system, the staff concludes that the design basis requirement of the containment spray cooling will not be affected.

3.2.10 Reactor Water Cleanup (RWCU) System

The RWCU system removes soluble and insoluble impurities from the reactor water, reducing their accumulation in the reactor coolant system. Maintaining low concentration of impurities prevents corrosion of reactor components and buildup of high radiation levels. The reactor water, after being introduced into the RWCU system, is first cooled in the regenerative and non-regenerative heat exchangers and then the insoluble and soluble impurities are removed by the filter-demineralizer. Purified water is then reintroduced into the coolant system through the regenerative heat exchanger. The temperature, pressure, and flow in the RWCU system will not change after the power uprate so that the integrity of the components in the system will remain unaffected. The only small change produced by the power uprate is a slight increase in the chemical impurity concentration factor for the reactor vessel caused by the increase of the feedwater flow. However, this is only a minor change which is not expected to significantly affect operation of the RWCU system. The staff reviewed and evaluated the licensee's analyses of the RWCU system performance, and, for the above reasons, concurs that it will not be affected by the power uprate.

3.2.11 Control Rod Drive (CRD) Hydraulic System

Structural Integrity

The licensee evaluated the effect of the proposed power uprate on the structural integrity of the CRD hydraulic system. The licensee stated that the reactor vessel operating and design pressure and temperature and the CRD hydraulic system pressure, temperature, and flow used in the existing design basis analysis remain bounding. The licensee concluded that the system will continue to perform its function and maintain its structural integrity at the proposed power uprate condition. Considering the large stress margin provided in the UFSAR, the staff agrees with the licensee's conclusion that the CRD hydraulic system will continue to meet the design basis and performance requirements for the proposed 1.4% power uprate.

Reactivity Control

The CRD hydraulic system controls gross changes in core reactivity by positioning neutron-absorbing control rods within the reactor. It is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The licensee concluded that the power uprate will not change the operating pressure, temperature, and core flow and that the CRD hydraulic system is, therefore, not affected by the power uprate.

The staff agrees that the proposed power uprate will not have a significant impact on the CRD hydraulic system for the following reasons:

(1) the operating dome pressure will not change, and the scram timing from steady state power conditions will not be affected;

(2) the proposed power uprate may minimally affect the scram timing during transient overpressure conditions; but, after the initial degraded scram time, the reactor pressure will assist the scram; and

(3) there must be a minimum pressure differential of 250 psid between the hydraulic control unit (HCU) and the vessel bottom head for normal CRD insertions and withdrawals. Again, since the operating dome pressure will not increase, the power uprate will have little impact on the CRD pump capacity.

Therefore, the staff agrees the CRD hydraulic system will continue to perform all its safety-related functions at the proposed uprated conditions.

3.2.12 Reactor Coolant System and Connected Systems - Conclusion

Based on the evaluation in Sections 3.2.1 through 3.2.11 of this safety evaluation, the staff concludes that the proposed power uprate is acceptable with respect to its impact on the reactor coolant system and connected systems.

3.3 Engineered Safety Features

3.3.1 Containment Systems

3.3.1.1 Containment Response Analysis

The licensee's application stated that the current licensing basis analyses for both short-term and long-term containment pressure and temperature following a LOCA assume an initial core power of 102% of current rated thermal power. Thus, the 1.4% power uprate is bounded by the original analyses.

In addition, the licensee preformed a separate suppression pool temperature analysis in accordance with NUREG-0738, "Suppression Pool Temperature Limits for BWR Containments," to demonstrate that local pool temperatures were within the condensation stability limits. The analysis used 104.3% of current rated power, which bounds the 1.4% power uprate.

Based on our review of the licensee's application, as set forth above, the staff agrees that the current licensing basis analyses and suppression pool temperature analysis are bounding and that containment response to a LOCA will not be affected by the 1.4% power uprate.

3.3.1.2 Containment Dynamic Loads

In 1984, the licensee submitted a plant-specific analysis of containment dynamic loads due to a LOCA with SRV blowdown to address requirements in NUREG-0661, "Mark I Containment Long Term Program." The analysis assumed an initial power level of 102% of current rated power to determine the design basis hydrodynamic loads. There is no change to the initial SRV set pressure as a result of the proposed 1.4% power uprate. However, for subsequent SRV actuations, the time between actuations may be slightly shorter. The licensee's analysis stated that the SRV hydrodynamic design basis ensures that subsequent actuations occur only after the water level oscillations have damped out and the level has stabilized at a point determined by the drywell-to-wetwell differential pressure minus the vacuum breaker setpoint. Primary system analyses are used to confirm that more than the minimum required time is available for the SRV discharge line water leg to return to the equilibrium position. To ensure that the water leg will be at equilibrium for all subsequent SRV actuations, there is delay logic on the two SRVs with the lowest pressure setpoints to allow the water leg to clear after the initial actuation. Thus, there is no impact on the SRV hydrodynamic loads, and the existing design for the LOCA hydraulic loads bounds the 1.4% power uprate.

Based on our review of the licensee's application, as set forth above, the staff agrees that the licensee's existing containment dynamic load analysis is bounding and will not be affected by the 1.4% power uprate.

3.3.1.3 Subcompartment Pressurization

The licensee analyzed the containment subcompartments, including the bioshield annulus and the drywell head. The licensee's RPV bioshield annulus area pressurization analysis considered both a reactor recirculation line break and a feedwater line break. The reactor pressure and temperature are not changed by the 1.4% power uprate, so the recirculation line break evaluation remains valid. The feedwater temperature and pressure parameters change slightly as a result of the 1.4% power uprate, but the original analysis remains valid because it assumed 105% of current design steam flow conditions, which bounds the slight changes in the feedwater parameters.

The licensee's application stated that the drywell head area pressurization analysis assumes a rupture of the RPV head spray piping. Since the reactor operating parameters are unchanged for the 1.4% power uprate, the analysis remains valid.

Based on our review of the licensee's application, as set forth above, the staff agrees that the licensee's existing subcompartment pressurization analysis is unchanged for the recirculation line break and the head spray piping rupture, and is bounding for the feedwater line break, and will not be affected by the 1.4% power uprate.

3.3.1.4 Containment Isolation

The licensee's application stated that since the current containment response analysis was based on 102% of current rated power level, the 1.4% power uprate will have no effect on the isolation valves connected to the reactor coolant pressure boundary or the balance of plant systems. In addition, the licensee reviewed the BWR motor-operated valve (MOV) program guidelines to ensure HCGS continues to conform to the guidance in Generic Letter 89-10, "Safety-Related Motor-Operated Valve (MOV) Testing and Surveillance." In both the main steam system and the feedwater system, the normal operating pressure, temperature, and/or flow experience a slight increase as a result of the 1.4% power uprate.

The main steam system will experience a slight increase in flow but no increase in pressure or temperature. Safety-related valves in the main steam system include the drain line isolation valves, main steam drain valves, steam header downstream drain isolation valve, startup drain valves, main steam stop valve and the air-operated MSIVs. Only the MSIVs will see an increase in flow rate, and since they are flow-assisted, the increase will help the closing function.

The feedwater system safety-related valves include the inlet check valves, and the crosstie isolation valve. There is no differential pressure across the inlet check valves for normal or abnormal operating conditions. The differential pressure across the crosstie isolation valve was calculated using the pump shutoff head and low water temperature to maximize the contribution of the pump head and the momentum head caused by the greater water density.

Based on our review of the licensee's application, as set forth above, the staff agrees that the containment isolation function of valves connected to the reactor coolant pressure boundary or BOP systems will not be affected by the 1.4% power uprate.

3.3.2 <u>ECCS</u>

Net Positive Suction Head (NPSH)

The ECCS is designed to mitigate design basis accidents, and includes the high pressure coolant injection (HPCI) system, the automatic depressurization system (ADS), the core spray system, and the LPCI system. The existing design basis analysis assumes an accident occurs at 104.2% of current rated power and 105% of current design steam flow. Therefore, the existing analysis bounds the 1.4% power uprate. The licensee calculated the NPSH for the ECCS pumps in accordance with Regulatory Guide 1.1, "NPSH for Emergency Core Cooling and Containment Heat Removal System Pumps." The suppression pool temperature analysis was performed based on 104.2% of current rated reactor thermal power, which bounds the proposed 1.4% power uprate.

Based on our review of the licensee's application, as set forth above, the staff agrees that the licensee's existing NPSH analysis is bounding and that the 1.4% power uprate will have an insignificant impact on the ECCS NPSH.

LOCA Analyses

The ECCS is designed to provide protection in the event of a LOCA due to a rupture of the primary system piping. Although design basis accidents (DBAs) are not expected to occur during the lifetime of a plant, plants are designed and analyzed to ensure that the radiological dose from a DBA will not exceed the 10 CFR Part 100 limits. For the LOCA, 10 CFR 50.46 specifies design acceptance criteria based on (1) the peak cladding temperature, (2) local cladding oxidation, (3) total hydrogen generation, (4) coolable core geometry, and (5) long-term cooling. The LOCA analysis considers a spectrum of break sizes and locations, including a rapid circumferential rupture of the largest recirculation system piping. Assuming a single failure of the ECCS, the LOCA analyses identify the break sizes that place the most severe challenge on the ECCS and the primary containment. The MAPLHGR operating limit is based on the most limiting LOCA analysis, and for each new fuel type, the licensee performs LOCA analyses to demonstrate that the 10 CFR 50.46 acceptance criteria can be met.

PSEG evaluated the impact of the power uprate on the performance of the ECCS in the event of a LOCA. The licensee stated that the HCGS ECCS provides adequate core cooling with the reactor operating at a bounding power and flow (104.2% of current rated power and 105% of current design core flow for two-recirculation-loop operation) when the event begins. The licensee concluded that the existing design basis LOCA analysis bounds the ECCS LOCA analysis at 101.4% of current rated power with 0.6% power measurement uncertainty.

The HCGS ECCS LOCA analyses were performed in accordance with NRC-approved methodologies and codes. The analyses demonstrate that the ECCS provides sufficient core cooling for a range of breaks with single ECCS system failures. Since the steady state operating pressure will not change and the rated steam flow will only change by 1.8%, the ECCS pumps' capability to inject or spray the assumed flow into the vessel during a LOCA will not be affected. In addition, the ECCS LOCA analyses of record for HCGS bound the uprated operating condition even with a 2% power measurement uncertainty (103.4% of current rated power). Therefore, the staff accepts the licensee's assessment that the ECCS will perform as designed and analyzed at the uprated conditions.

ADS

The ADS uses the SRVs to reduce reactor pressure following a small-break LOCA with HPCI failure, allowing LPCI and core spray to provide cooling flow to the vessel. The licensee stated that the ADS initiation logic and ADS valve control are adequate for the power uprate. The plant design requires a minimum flow capacity for the SRVs, and after a time delay, the ADS initiates either on low water level plus high drywell pressure or on low water level alone. The licensee stated that the ability to perform either of these functions is not affected by the power uprate and the staff concurs.

3.3.3 ECCS Performance Evaluation

The ECCS is designed to provide protection against hypothetical LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. PSEG stated that since the changes to the core operating parameters are small, the NRC-approved methodology specified in CENPD-300-P-A remains applicable in the accident analyses. The licensee stated that the LOCA analyses of record used initial condition inputs based on 104.2% of current rated power and 105% of current design core flow and that the rod heatup analysis is also based on 102% of the current rated thermal power. Therefore, the LOCA analyses remain applicable at the uprated conditions.

Topical report CENPD-300 describes HCGS's accident analysis methodologies. This NRC-approved topical report also specified the related NRC-approved topical reports and the codes for the ECCS safety analyses. The staff accepts the HCGS ECCS performance evaluation because the analytical model is based on an NRC-approved methodology and the analyses are based on bounding power and flow conditions.

3.3.4 Post-LOCA Combustible Gas Control

Hydrogen recombiners are used following a LOCA to maintain containment atmosphere hydrogen levels below combustible levels. The current licensing basis analysis assumes a LOCA occurs at 104.5% of current rated thermal power, which bounds the proposed 1.4% power uprate.

Based on our review of the licensee's application, as set forth above, the staff agrees that the licensee's existing analysis bounds the 1.4% power uprate and its impact on the hydrogen recombiners is negligible.

3.3.5 Filtration Recirculation and Ventilation System (FRVS)

The FRVS removes fission products from the reactor building enclosure atmosphere following an accident that results in a release of radioactivity in either the primary containment or the reactor building (secondary containment). It also maintains a negative pressure of approximately 0.25 inch water gauge in the reactor building to minimize any unfiltered release of fission products. The existing licensee analysis assumes a reactor power level of 102% of current rated power. Therefore, operation of the FRVS is not affected by the 1.4% power uprate.

Based on our review of the licensee's application, as set forth above, the staff agrees that the licensee's existing analysis for the FRVS remains valid for the 1.4% power uprate.

3.3.6 Control Room Heating, Ventilation and Air Conditioning (HVAC) Systems

The control room HVAC system includes the control room emergency filter (CREF) system and is designed to maintain the control room at a slightly positive pressure to minimize unfiltered in-leakage following a design basis accident. The current radiological assessment for control room habitability assumes a power level of 102% of rated thermal power. This assessment is bounding for the proposed 1.4% power uprate.

Based on our review of the licensee's application, as set forth above, the staff agrees that the licensee's existing analysis for the control room HVAC and CREF systems is bounding for the 1.4% power uprate.

3.3.7 Engineered Safety Features - Conclusion

Based on the evaluation in Sections 3.3.1 through 3.3.6, the staff concludes that the proposed power uprate is acceptable with respect to its impact on the HCGS Engineered Safety Features.

3.4 Instrumentation and Controls

3.4.1 Instrumentation and Controls - Background

As previously discussed, the licensee requested approval to increase the HCGS licensed thermal power level based on the installation of the CENP Crossflow UFM system. The Crossflow system is designed to improve the accuracy of feedwater flow rate measurement, which is used, in part, to calculate reactor thermal power. The improved flow measurement instrumentation would allow PSEG to operate HCGS with a reduced margin between the actual power level and the 102% power level previously used in the licensing basis ECCS analyses.

The theory, design, and operating features of the Crossflow system are documented in CENP topical report CENP-397-P-A, Revision 01, “Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology,” dated May 2000. The topical report was approved by the NRC in a safety evaluation dated March 20, 2000.

The typical elements used for measuring feedwater flow are an orifice plate, a venturi meter, or a flow nozzle, which generate a differential pressure proportional to the feedwater velocity in the pipe. Of the three differential pressure devices, the venturi meter is most widely used for feedwater measurement in nuclear power plants. The major advantage of a venturi meter is a relatively low head loss as the fluid passes through the device. However, fouling of the device is the major disadvantage of this meter or any other nozzle-based flow meter. Fouling is a metallic plating on the throat area of the meter, which causes the meter to indicate a higher differential pressure and thus a higher than actual flow rate. This result leads plant operators to calibrate nuclear instrumentation high. Calibrating nuclear instrumentation high is conservative with respect to reactor safety, but causes the electrical output to be proportionally low when the plant is operated at its thermal power rating. In addition to fouling, two instruments (the transmitter and the analog-to-digital converter of the venturi meter), introduce errors in the flow measurement, necessitating removal, cleaning, and recalibration of the flow device. Because of the desire to improve flow instrumentation uncertainty and to operate the plant closer to the licensed power rating, the industry assessed alternative flow measurement techniques and found the UFM to be a viable alternative. The UFM is an electronic transducer that is controlled by computer and is not susceptible to fouling because it does not have differential pressure elements.

The Crossflow UFM is a cross-correlation UFM system. A cross-correlation UFM is a clamp-on (clamp-on refers to the method of attaching ultrasonic transducers used in the flow measuring process by means of a clamp device mounted external to the pipe) UFM. The cross-correlation technique determines the velocity of the fluid by measuring the time taken by a unique pattern of eddies in the fluid to pass between two sets of transducers, each set at a certain known distance apart, injecting ultrasonic signals perpendicular to the pipe axis. The cross-correlation UFM was first developed by the Canadian General Electric for Ontario Hydro. However, the system was not optimized for application over a wide range of flow velocities and pipe diameters. The task of optimizing the cross-correlation technique was carried out by the Advanced Measurement Analysis Group, Inc., and CENP. This work resulted in an improved cross-correlation flow meter called "Crossflow." These new flow meters have been installed to measure reactor coolant flow and steam generator feedwater flow in more than 40 nuclear power plants in the United States, Canada, South America, and Europe. However, licensees have not taken credit for the Crossflow UFM in regulatory applications.

3.4.2 Instrumentation and Controls - Evaluation

PSEG will install a Crossflow UFM system for feedwater flow measurement at HCGS. This system consists of four ultrasonic transducers (two transmitters and two receivers) that are mounted in a saddle-type support frame, which is externally attached to the pipe in which the flow is to be measured. The ultrasonic transducers are connected to a signal conditioning unit (SCU) and a data processing computer (DPC). The carbon steel saddle-type support frame has transducer holes that are bored in one run by a computerized numerical control machine. This frame provides an exceptionally accurate alignment of the transducers, and no field adjustments are needed. The DPC, with its Crossflow software, performs digital signal processing on the demodulated ultrasonic signals and calculates the delay time for use in the flow calculation. The Crossflow software verification and validation are performed in accordance with the CENP quality and implementing procedures manuals.

CENP Topical Report CENPD-397-P-A, Revision 01, provides information on the Crossflow UFM system design, its underlying principles of ultrasonic measurement, experimental data validating system accuracy, and an overview of installation. The combined effect of these elements is to provide an improvement in flow measurement accuracy over current flow measurement systems. This increased flow accuracy can be translated into a like improvement in the accuracy of the core power level calculation, due to the use of a more accurate feedwater flow in the heat balance calculation. The measurement uncertainty reduction allows a utility to: (1) operate the plant at a higher power level without exceeding the 10 CFR Part 50, Appendix K, mandated 102% power level (2% margin attributed to instrument uncertainty); (2) apply the reduced uncertainty to overall margin improvement; (3) recover lost generating capacity due to feedwater venturi fouling while staying within the plant’s licensed operating power level; and (4) possess an in-plant capability for periodically recalibrating the feedwater venturi flow coefficient to adjust for the adverse effect of fouling.

Topical Report CENPD-397-P-A, Revision 01, provided a methodology for determining measurement uncertainty of the Crossflow UFM. This methodology uses specific guidelines and equations for determining uncertainty values of the Crossflow input parameters with a 95% confidence interval. The parameters that contribute to feedwater flow measurement uncertainty are pipe inside diameter, transducer spacing, feedwater density, Crossflow time delay, pipe wall roughness, and the velocity profile correction factor (VPCF). CENPD-397-P-A, Revision 01, included typical uncertainties for each of the input parameters, except for the pipe wall roughness, and the overall flow measurement uncertainty of the Crossflow UFM for a typical feedwater loop (straight pipe, fully developed flow). Actual uncertainties are determined on a plant-specific basis by using the guidelines and equations provided in the topical report. Most of the uncertainties are affected by temperature change. Therefore, the topical report recommended improving the accuracy of the feedwater temperature instrumentation to reduce the total uncertainty of the feedwater flow measurement. The accuracy of the Crossflow time delay is confirmed monthly in the field for a specified acceptance value, and the power plant licensee is advised in the topical report to accurately measure the feedwater temperature. The methodology specified additional correction factors to be applied to the VPCF of a fully developed flow in a straight pipe when determining the VPCF for plant-specific conditions and pipe configurations.

The December 1, 2000, submittal provides a justification for a 1.4% power uprate by using the Crossflow UFM system to determine the plant thermal power. The Crossflow uncertainty calculation indicated a mass flow accuracy of better than 0.5% of rated flow for HCGS. The calculations are consistent with the methodology described in Topical Report CENPD-397-P-A, Revision 01. The Crossflow uncertainty calculation supports an uncertainty in the reactor power measurement of 0.6% with a 95% confidence level. The submittal lists the contributions of individual error elements and states that the calculations are based on an accepted plant instrument uncertainty methodology. The calculation of total power measurement uncertainty of the venturi and the Crossflow UFM used the root sum square method to combine the individual error elements (various error elements related to Crossflow UFM, pressure and moisture instrument errors, and other gains and losses).

The licensee's application stated: "Although use of the Crossflow system for this application is non-safety-related, the system is designed and manufactured under the vendor's quality control program which provides for configuration control, deficiency reporting and correction, and maintenance. The current software was verified and validated under CENP's Verification and Validation Program. Specific examples of quality measures included in the design, fabrication and testing of the Crossflow system are provided in the topical report. CENP's Verification and Validation [V&V] program provides procedures for deficiency reporting for engineering action and notification of holders of V&V software."

The licensee's application also stated: "The Crossflow system will be included in the plant preventive maintenance program. Technical support personnel will monitor the Crossflow system's reliability. Equipment problems will be documented and corrected in accordance with PSEG Nuclear's corrective action program. Conditions that are adverse to quality are documented under the corrective action program. The system software is subject to PSEG Nuclear's software quality assurance program."

The staff's safety evaluation on CENP Topical Report CENPD-397-P, Revision 01-P, included four additional criteria to be addressed by a licensee requesting a power uprate. In the December 1, 2000, submittal, PSEG addressed each of the four criteria as follows:

(1) *The licensee should discuss the development of maintenance and calibration procedures that will be implemented with the Crossflow UFM installation. These procedures should include processes and contingencies for an inoperable Crossflow UFM and the effect on thermal power measurement and plant operation.*

In Attachment 1 of the December 1, 2000, submittal, PSEG stated: "Crossflow system failures are detected and transmitted to the plant computer which causes an overhead annunciator point to alarm for Crossflow abnormal conditions so that the operators are aware of Crossflow status. The Crossflow system does not perform any safety function and is not used to directly control any plant systems. Therefore, system inoperability has no immediate effect on thermal power measurement uncertainty or plant operation.

If the Crossflow system becomes unavailable, plant operation at the core thermal power level of 3339 MWt may continue for 24 hours after the last valid correction factor was obtained from the Crossflow system. Procedural guidance would direct that reactor power be reduced to a level less than or equal to the previously licensed power level (3293 MWt) if the Crossflow system cannot be restored to operation within 24 hours. Core power would be maintained at a level less than or equal to 3293 MWt until the Crossflow system was returned to service and a heat balance in accordance with SR [Surveillance Requirement] 4.3.1.1 was performed with updated factors from the Crossflow system.

Calibration and maintenance of the Crossflow system will be performed using site procedures developed from the Crossflow system technical manuals. All work is performed in accordance with site work control procedures. Verification of Crossflow system operation is provided by onboard system diagnostics.

Crossflow operation will be monitored on a periodic basis using an internal time delay check. In this way, the user is able to verify that the SCU, DPC, and software remain within the stated accuracy."

In the May 7, 2001, response to the staff's April 9, 2001, request for additional information, PSEG described its programs for the calibration of all other instrumentation, in addition to the Crossflow, whose measurement uncertainties affect the plant power uncertainty. These other instruments are for feedwater temperature, reactor pressure, control rod drive flow, reactor water cleanup flow, reactor water cleanup temperature, and recirculation pump watts. The licensee identified plant procedures applicable to these instruments and specified their calibration intervals. The licensee also listed PSEG's applicable procedures for performing corrective actions, reporting deficiencies to the manufacturers, and receiving and addressing manufacturer deficiency reports on these instruments. The staff believes that the licensee's plant procedures can assure instrumentation capability sufficient to provide acceptable power uncertainty for the proposed power uprate.

(2) *For plants that currently have Crossflow UFM installed, the licensee should provide an evaluation of the operational and maintenance history of the installed UFM and confirm that the instrumentation is representative of the Crossflow UFM and is bounded by the requirements set forth in Topical Report CENPD-397-P-A, Revision 01.*

In Attachment 1 of the December 1, 2000, submittal, PSEG stated: "The Crossflow system will be installed before implementation of the proposed uprate. Therefore, plant-specific maintenance and operations data is not available for evaluation. However, significant operational experience has been accumulated from installations at several nuclear power plants. The cumulative operating history shows that the Crossflow system has proven to be reliable. To date, excluding dryout of a couplet that will not be used at Hope Creek, no Crossflow installations have experienced failures which adversely impact the ability to provide the venturi recalibration function. This is over a period of approximately 136 effective years of operational flow measurements.

The Crossflow system that will be installed at Hope Creek is representative of the Crossflow UFM of Topical Report CENPD-397-P-A, Revision 01 and is bounded by the requirements set forth in the topical report."

(3) *The licensee should confirm that the methodology used to calculate the uncertainty of the Crossflow UFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternate methodology is used, the application should be justified and applied to both the venturi and the Crossflow UFM for comparison.*

In Attachment 1 of the December 1, 2000, submittal, PSEG stated: "CENP has completed the Hope Creek Crossflow uncertainty calculation indicating a mass flow accuracy of better than 0.5% of rated flow for the site specific installation. The calculations are consistent with the methodology described in Topical Report CENPD-397-P-A, Revision 01. The uncertainty calculations specify requirements for 95% confidence interval flow measurement including:

- Inside pipe diameter measurement and associated uncertainty,
- Transducer spacing measurement and associated uncertainty,
- Velocity Profile Correction Factor (VCPF) and justification, and
- Crossflow time delay calibration data and associated uncertainty

The Crossflow uncertainty calculation supports an uncertainty in the reactor power measurement of 0.6% as shown in Attachment 7. The uncertainty is at a 95% confidence level (2σ). These calculations are based on accepted plant instrument uncertainty methodology.

Crossflow system operating procedures will ensure the assumptions and requirements of the uncertainty calculation remain valid."

(4) *The licensee of a plant at which the installed Crossflow UFM was not calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation) should provide additional justification. This justification should show either that the meter installation is independent of the plant-specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibration and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated Crossflow UFM, the licensee should confirm that the plant-specific installation follows the guidelines in the Crossflow UFM topical report.*

In Attachment 1 of the December 1, 2000, submittal, PSEG stated: "The Hope Creek Crossflow installation will be installed and calibrated to a site-specific piping configuration (flow profile and meter factors are representative of the plant-specific installation). The installation follows the guidelines in the Crossflow UFM topical report."

The staff finds that PSEG's response to these criteria has resolved the plant-specific concerns about Crossflow UFM maintenance and calibration, hydraulic configuration, processes, and contingencies for an inoperable Crossflow UFM. The licensee used an approved methodology to calculate the plant-specific Crossflow measurement uncertainty and the power measurement uncertainty.

3.4.3 Instrumentation and Controls - Conclusion

Based on the evaluation in Section 3.4.2 of this safety evaluation, the staff finds that the calculation of the power measurement uncertainty for the HCGS power uprate is acceptable. Based on the staff's review of the plant-specific calculation, the staff finds that the HCGS Crossflow UFM thermal power measurement uncertainty is limited to 0.6% of actual reactor thermal power and can support the proposed 1.4% uprate of the HCGS licensed thermal power. The staff also finds that the licensee sufficiently addressed the four additional criteria outlined in the staff safety evaluation on CENP Topical Report CENPD-397-P, Revision 01-P.

3.5 Electrical Systems

3.5.1 Electrical Systems - Background

The electric power system is designed to generate and transmit electric power into the Pennsylvania-New Jersey-Maryland (PJM) power grid. The offsite power for the plant is fed through the 500 kV system via the 13.8 kV yard ring bus. The 13.8 kV ring bus is the preferred source of auxiliary power during startup, normal operation, and shutdown. In the event of a loss of offsite power, four independent standby diesel generators provide power for Class 1E loads and selected non-Class 1E loads. The generator unit uses a three single-phase power transformer arrangement that steps up the voltage from 25 kV to 500 kV.

Grid stability and availability is maximized because the utility is a member of the PJM interconnected network. The bulk system transmission facilities were tested using system simulation for both power flow and transient stability studies. The stability analysis was conducted for the system configuration using the Load Flow and Dynamic stability program provided by Power Technologies Inc.

For the proposed change, the licensee used the guidelines described in NEDO-31897, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," issued in February 1992. The following sections provide the evaluation of the proposed power uprate on electrical systems, structures, and components.

3.5.2 Electrical Distribution System

The licensee does not expect additional demands for the ac or dc auxiliary loads because of this power uprate. Therefore, the plant auxiliary ac and dc electrical load will not change. The licensee evaluated the voltages and short circuit currents at different levels of the station auxiliary electrical distribution system and determined that these will not change as a result of uprate because the uprate capacity remains within the main generator rating, and the voltage controls and grid impedances at the PJM 500 kV grid will not be affected.

3.5.3 Turbine-Generator

The HCGS steam-turbine-driven generator is a 4-pole machine, rated at 1,300 MVA at a 0.9 power factor. This rating is based on 75 psig hydrogen pressure, which is supplemented with water cooling for the stator and rotor. At the current thermal rating of 3,293 MWt, the main generator gross electrical output is 1,118 megawatts electric (MWe). The licensee has committed to revise the generator capability curves, ensuring that the anticipated net increase of 15 MWe will remain well within the limits of the generator. The licensee has not identified any changes to relay settings for the generator, although some process alarm setpoints for the generator and exciter may require adjustment. The electrical system associated with turbine auxiliary systems is within the design rating and is not affected by the uprate.

3.5.4 Isophase Bus

The isophase bus is designed with a forced cooling rating of 32,000 amperes. This rating is greater than the main generator rating of 30,022 stator amperes at 1,300 MVA and exceeds the anticipated generator output. The isophase bus will support the power increase with no modifications.

3.5.5 Main Power Transformers

The licensee’s application stated that system operating procedures will be revised as required to ensure that operation of the generator remains within applicable limits for the main power transformers at the 1.4% uprated power.

3.5.6 Station Service Transformers

The station service transformers receive power from 13.8 kV ring bus and supply the electrical distribution system. The station service transformers are sized for full loading of the plant equipment. Since no change to the electrical components is required for the uprating, the adequacy of the station service transformers continues to be assured.

3.5.7 Switchyard

The licensee’s application stated that all 500 kV equipment was designed for transmission of the 1,300 MVA generator rating at unity power factor. The switchyard will accept the additional load without the need for any hardware modifications.

3.5.8 500 kV Grid Stability

The licensee did not identify any stability issues during a feasibility study performed in support of the proposed uprate. Although the staff is not aware of any such stability issues, the staff notes that the licensee has committed to do an impact study, including a stability analysis, before implementation of the proposed change.

3.5.9 Electrical Systems - Conclusion

Based on the evaluation in Sections 3.5.2 through 3.5.8 of this safety evaluation, the staff concludes that the licensee has provided reasonable assurance that the safety functions of the electrical power system will be maintained and will have a negligible impact on grid stability. This is consistent with GDC 17 and the proposed change is, therefore, acceptable.

3.6 Auxiliary Systems

3.6.1 Fuel Pool Cooling

The fuel pool cooling and cleanup system (FPCCS) removes heat from the spent fuel assemblies stored in the spent fuel pool in order to maintain the pool temperature at or below its design temperature during normal plant operations. In addition, the FPCCS reduces activity, maintains water clarity, and maintains the cooling function during and after a seismic event.

The FPCCS is designed to maintain the spent fuel pool temperature at 135 °F under a design heat load of 16.1E+6 Btu/hr, which is based on 16 consecutive refuelings with one-third of the core removed during each refueling, at a refueling frequency of 18 months. The RHR system can be used to supplement the FPCCS through a crosstie that can maintain the pool temperature at 150 °F with a heat load of 34.2E+6 Btu/hr, which is based on a full-core offload at the end of a fuel cycle plus the decay heat of 13 previous refuelings of one-third of the core. Although partial core offloads are the common practice at HCGS, full-core offloads are performed from time to time and are not considered to be emergency procedures. The licensee reviewed the spent fuel heat load calculations and found that there is adequate margin for the slight increase in decay heat load (proportional to the 1.4% power uprate).

Based on our review of the licensee’s application, the staff agrees that the FPCCS, in combination with the RHR system, can maintain the spent fuel pool temperature at or below design limits for all core offload conditions at the proposed 1.4% uprated power level.

The licensee’s application stated that the proposed power uprate will not cause a significant change in the amount of impurities introduced into the spent fuel pool. Therefore, the capability of the FPCCS to maintain pool purity and clarity is not expected to be affected. The staff concurs with the licensee’s evaluation.

3.6.2 Cooling Water Systems

The station service water system (SSWS) provides water from the ultimate heat sink (the Delaware river) to the heat exchangers of the safety auxiliary cooling system (SACS) and the reactor auxiliary cooling system (RACS). The SSWS heat removal capability under accident and transient conditions was analyzed for 102% of the current licensed power level, which bounds the 1.4% uprate. During normal operations, the SACS heat exchangers service both the SACS and the turbine auxiliary cooling system (TACS), but during a loss of offsite power or a LOCA, the TACS is isolated so that only the SACS is serviced by the SACS heat exchangers in order to cool safety-related equipment. The RACS is also isolated during design basis accidents so that the SSWS provides cooling only to the SACS heat exchangers.

The RACS is a closed loop system that provides cooling to non-safety-related components, including the reactor recirculation pump seal and motor oil cooler, reactor water cleanup system pump seal cooler, nonregenerative heat exchanger, control rod drive pump seal cooler, and miscellaneous condensers and coolers.

The TACS provides cooling to the non-safety-related turbine auxiliary components, including the turbine-generator cooling equipment, turbine building chillers, instrument air compressors, reactor feed pump turbine oil, condensate pump motor bearings, and miscellaneous non-safety equipment and room coolers.

The SACS provides post-accident decay heat removal by providing cooling water to safety-related components. The system was sized to mitigate the consequences of accidents or transients, assuming an initial power level of 102%. Therefore, the original design basis heat removal capacity bounds the 1.4% power uprate conditions.

Since the RACS and TACS do not perform any safety-related function, the impact of the proposed uprated power operations on the designs and performances of these systems was not reviewed.

Based on our review of the licensee’s application, as set forth above, we find that plant operations at the proposed 1.4% uprated power level do not change the design aspects and operations of the SSWS or the SACS systems. Therefore, the staff agrees with the licensee's conclusion that the impact of plant operations at the proposed uprated power level on these systems is insignificant.

3.6.3 Standby Liquid Control (SLC) System

The function of the SLC system is to provide the capability of bringing the reactor from full power to a cold xenon-free shutdown, assuming that none of the withdrawn control rods can be inserted. The Anticipated Transients Without Scram (ATWS) rule (10 CFR 50.62) requires, in part, that each BWR have a SLC system with the capability of injecting into the reactor pressure vessel a borated water solution such that the resulting reactivity control is at least equivalent to the control obtained by injecting 86 gpm of a 13-weight-percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside-diameter reactor vessel.

The licensee stated that the SLC system can inject sodium pentaborate, which would produce an equivalent concentration of at least 660 ppm of natural boron in the reactor. The licensee added that at HCGS, an additional 25% concentration (825 ppm) is maintained to ensure the reactivity control capability of the SLC system. The boron concentration required to ensure SLC shutdown capability is evaluated for each core reload.

The SLC system is designed to inject at a maximum reactor pressure equal to the minimum SRV setpoint pressure. The licensee stated that the SRVs begin to lift at approximately 1,100 psig and the maximum SRV lift setpoint is 1,130 psig ±3%. The SLC pumps are positive displacement pumps and the licensee stated that the pumps have sufficient margin up to the system relief valve setting of 1,400 psig. The licensee stated that the reactor operating pressure and the nominal SRV setpoints are not changed in the proposed power uprate.

According to the licensee, the SLC pumps will continue to function as designed and the SLC capability to provide backup shutdown and the ATWS functions are not affected by the proposed power uprate. Based on our review of the licensee's application, as set forth above, the staff agrees with the licensee's conclusion that the SLC system operation will not be affected by the proposed power uprate.

3.6.4 HVAC Systems

The function of the HVAC systems is to prevent extreme thermal environment conditions from affecting personnel and equipment by ensuring that design temperatures are not exceeded. HVAC systems that could potentially be affected by the requested power uprate include drywell cooling, reactor building HVAC, turbine building HVAC, auxiliary building service, and radwaste area ventilation.

The licensee's application stated that the reactor vessel operating temperature and design recirculation flow are not affected by the power uprate. There is a small increase of about 1 °F (to 421 °F) at the outlet of the sixth feedwater heater, but the original piping heat load was calculated using 425 °F based on a turbine valves-wide-open heat balance, so the original calculation is bounding. In addition, a 20% margin was added to the calculated drywell heat loads. Therefore, the power uprate will have no impact on drywell cooling.

During normal operations the reactor building HVAC and its subsystems supply the reactor building areas, including the ECCS pump rooms, the refuel floor, the RWCU system equipment, and the steam tunnel. Post-accident heat loads in the ECCS pump rooms will not change as a result of the 1.4% power uprate because the original containment response analysis was based on 102% of the current rated power. The heat loads for the refuel floor and RWCU were also originally calculated at the conservative 102% power level. The 1°F increase in feedwater/condensate temperature is bounded by the original heat load calculations performed at 102% power. Thus, the reactor building HVAC, including the steam tunnel cooling, will not be affected by the power uprate.

The turbine building HVAC heat load calculation includes a 15% margin above the calculated heat loads. Therefore, it is not expected to be affected by any slight increase in heat loads from the piping and turbine-generator cooling systems.

Based on our review of the licensee's application, as set forth above, the staff agrees with the licensee that plant operations at the proposed uprated power level has an insignificant or no impact on the HVAC systems for the above-cited areas.

3.6.5 Fire Protection

Fire suppression or detection is not expected to be affected by plant operations at the proposed 1.4% uprated power level since physical plant configurations and combustible loads will not change as a result of the uprated power operations. In addition, the safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change for the uprated conditions, and the operator actions required to mitigate the consequences of a fire are not affected.

Based on our review of the licensee's application, as set forth above, the staff agrees with the licensee that the safe shutdown systems and procedures used to mitigate the consequences of a fire will continue to meet 10 CFR 50.48 and 10 CFR Part 50, Appendix R, and will not be affected by plant operations at the proposed 1.4% uprated power level.

3.6.6 Chemistry and Radiochemistry Systems

Reactor water chemistry at HCGS is controlled in order to provide corrosion protection and minimize radiation levels in the plant. The chemistry control systems consist of condensate pre-filter and demineralizer, RWCU, hydrogen water chemistry, and zinc and iron injection systems. The licensee evaluated the effect of the power uprate on the performance of these systems. The function of the condensate pre-filter and demineralizer is to remove impurities from the reactor water. The licensee's analysis indicates that the power uprate will not significantly change the flow of the condensate and the condensate temperature and pressure will remain essentially unchanged. Therefore, the function of the condensate pre-filter and demineralizer will not be affected by the power uprate. The function of the RWCU system is to reduce insoluble and soluble impurities within the reactor coolant system. The licensee's application stated that the power uprate will not cause any significant change in the operation of this system. Since there is no change in the water chemistry, the effect of zinc and iron injection on minimizing plant radiation fields will not be affected. Due to the increase of radiolytic decomposition of reactor water after the power uprate, concentration of oxygen in the coolant may slightly increase. However, in the hydrogen water chemistry regime the 1.4% power increase can produce only a minimal increase in oxygen concentration. The increase is too small to have any significant effect on the corrosive effect of the reactor water components. The staff reviewed the licensee's evaluation of the reactor power uprate on the chemistry and radiochemistry systems. Based on our review of the licensee's application, as set forth above, the staff concludes that the 1.4% reactor power increase will not significantly change the performance of these systems.

3.6.7 Auxiliary Systems - Conclusion

Based on the evaluation in Sections 3.6.1 through 3.6.6 of this safety evaluation, the staff concludes that the proposed power uprate is acceptable with respect to its impact on the auxiliary systems.

3.7 Steam and Power Conversion Systems

The HCGS power conversion systems and their support systems (including the condenser air removal, steam jet air ejectors, turbine steam bypass, and condensate system) are designed for 105% of rated steam flow, which is the flow condition for turbine valves-wide-open. The licensee's application stated that the proposed 1.4% power uprate will increase the rated steam and feedwater flows by 1.8%, which is bounded by the 105% design value. Therefore, the licensee concluded that the proposed power uprate has no impact on the power conversion systems.

The staff issued a request for additional information (RAI) dated April 9, 2001, that, in part, involved the design analysis of the main turbine. Specifically, the staff asked if the analysis bounds the turbine overspeed and associated missile production for the 1.8% increase in steam flow. The licensee response, dated May 7, 2001, stated that the turbine overspeed and the associated missile production evaluation was performed at the turbine valves-wide-open condition (105% of steam flow), which bounds the 1.8% increase in steam flow resulting from the 1.4% power increase.

Based on our review and the licensee's response to the RAI, the staff agrees with the licensee that operation of the turbine at the proposed uprated power level is bounded by the original design of 105% steam flow for valves-wide-open. Thus, there is no increase in the probability of producing a turbine overspeed or an associated turbine missile at the 1.4% uprated power level.

Since the support systems do not perform any safety-related function, we have not reviewed the impact of plant operations at the proposed uprated power level on the design and performance of these systems.

Based on the above evaluation, the staff concludes that the proposed power uprate is acceptable with respect to its impact on the steam and power conversion systems.

3.8 Radioactive Waste Management

3.8.1 Liquid Radwaste Management

The liquid radwaste management system collects, processes, monitors, and recycles or disposes of radioactive liquid wastes. The system was conservatively designed to process liquid waste based on power operation at 105% of current rated power (3,458 MWt). The licensee concluded that no changes to the liquid radwaste management system are required and that it is not affected by the 1.4% power uprate.

Based on our review of the licensee's application, as set forth above, the staff agrees with the licensee's conclusion and finds that the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, applicable to the liquid radwaste management system, will continue to be satisfied at the proposed 1.4% increase in power level.

3.8.2 Gaseous Radwaste Management

The gaseous radwaste management systems include all systems that process potential sources of airborne releases of radioactive materials during normal operation and anticipated abnormal operational occurrences. The gaseous radwaste management systems include the offgas system and various ventilation systems that reduce radioactive gaseous releases from the plant by filtration or delay to allow the decay of radioisotopes prior to release. The gaseous source terms for HCGS are calculated based on 105% of current rated power (3,458 MWt). Therefore, the licensee concluded that the original calculations are bounding and the gaseous radwaste management systems are not affected by the proposed 1.4% power uprate.

Based on our review of the licensee's application, as set forth above, the staff concludes that plant operations at the proposed 1.4% uprated power level will have an insignificant impact on the gaseous radwaste management system.

3.8.3 Radioactive Waste Management - Conclusion

Based on the evaluation in Sections 3.8.1 and 3.8.2 of this safety evaluation, the staff concludes that the proposed power uprate is acceptable with respect to its impact on radioactive waste management.

3.9 Radiological Consequences

The licensee performed an assessment of the radiological consequence analyses for a power uprate to 3,339 MWt in the submittal dated December 1, 2000. The licensee stated that the current radiological consequence analyses for the design basis loss-of-coolant accident, fuel handling accident, and control rod drop accident are based on a reactor core thermal power level of 3,458 MWt (105% of current rated thermal power level of 3,293 MWt); therefore, the 1.4% power uprate to 3,339 MWt is bounded by the current analyses.

The staff reviewed the radiological consequences analyzed for the above design basis accidents in Chapter 15 of the HCGS UFSAR and confirmed that the currently analyzed power level bounds the proposed uprate power level of 3,339 MWt, and that the radiological consequences calculated at a reactor core thermal power level of 3,458 MWt met the relevant dose acceptance criteria. Since the reactor accident source terms and release rates used in the current analyses are based on a reactor power level of 3,458 MWt, the calculated radiological consequences in the UFSAR remain bounding. The licensee proposed no change in methodology or assumptions for the radiological consequence assessments in its submittal.

The licensee’s submittal further stated that: (1) the source terms used for the radiological consequences resulting from the instrument line pipe break and steam system piping break accidents are based on a core thermal level of 3,435 MWt, which is 104% of the current thermal power, bounding the proposed 1.4% power uprate, and (2) the source terms used for the gaseous radwaste subsystem failure are based on a core thermal power level of 3,400 MWt, which also bounds the proposed 1.4% power uprate. The licensee stated that, therefore, the current radiological consequence assessments in the HCGS UFSAR also remain valid for the proposed core thermal power uprate.

For the steam system piping break and instrument line pipe break, the only radionuclides available for release resulting from these breaks are fission products that are present in the reactor coolant and steam lines prior to the breaks and those resulting from accident-initiated iodine spikes following the breaks. No fuel damage will directly result from these breaks. Therefore, the radiological consequences of these breaks are not affected by the proposed 1.4% power uprate.

The staff has reviewed the licensee’s submittal and has concluded that the current design basis dose analyses, as documented in the HCGS UFSAR, remain acceptable for the proposed 1.4% reactor core thermal power uprate. Therefore, the staff concludes that the proposed 1.4% power uprate is acceptable with respect to the radiological consequences of design basis accidents.

3.10 Human Factors

3.10.1 Plant Procedures

The licensee stated in the submittal dated December 1, 2000 that "[p]lant procedures will not require significant changes for the uprate. The same steps and sequences of steps will be maintained. The only new procedures required are for operation and maintenance of the Crossflow system." The licensee further stated that operations personnel will be trained on the revised plant procedures before the proposed Crossflow system is implemented.

The staff finds that the licensee's submittal is satisfactory because the licensee has adequately identified the type and scope of plant procedures that will be affected by the uprate and indicated that the procedures will be appropriately revised. Furthermore, the response indicated that operators will be trained on the changes before the procedures are implemented and adequately described the effect of the procedure changes on operator actions.

3.10.2 Risk-Important Operator Actions Sensitive to Power Uprate

The licensee stated in the submittal dated December 1, 2000, that "ESF System design and setpoints, and procedural requirements already bound the proposed uprate. The uprate will not change the time available for the operators to respond, or add additional steps."

The staff finds that the licensee's submittal is satisfactory because the licensee has adequately addressed the question of operator actions sensitive to the power uprate by describing the effect of the power uprate on operator performance and adequately justified the effect or lack of effect on required operator response.

3.10.3 Control Room Controls, Displays, and Alarms

In its December 1, 2000, submittal, the licensee stated that there will be minimum impact on alarms, controls, and displays for a 1.4% uprate. Indicated power (i.e., reactor power 100%) will be rescaled to indicate the new uprated power. Therefore, the increased megawatt rating will indicate at 100% power. Installation of the Crossflow UFM systems used for feedwater flow measurements will result in the subsequent installation of additional alarms and an additional annunciator tile in the main control room to alert the operators to conditions of UFM inoperability or inaccuracy. The licensee stated that alarms will be recalibrated as required to reflect setpoint changes though no significant or fundamental setpoint changes are anticipated. The licensee also stated that operator response to existing alarms is anticipated to remain as before.

The staff finds the licensee's submittal satisfactory because the licensee has adequately identified the changes that will occur to alarms, displays, and controls as a result of the power uprate and adequately described how these changes will be accommodated.

3.10.4 Safety Parameter Display System

The licensee's submittal stated that, "[p]rocess parameter scaling changes will be made as required for the Safety Parameter Display System (SPDS). No other changes to the SPDS are anticipated, and the scaling changes made will be controlled under PSEG Nuclear's software configuration change control program."

The staff finds the licensee's submittal satisfactory because the licensee has adequately identified the changes that will occur to the SPDS as a result of the power uprate and adequately described how the changes will be addressed.

3.10.5 Operator Training Program and the Control Room Simulator

The licensee indicated that "[s]ince the power uprate is nominal and there is no change to how the plant will be operated, the impact on operator training is minimal." The licensee also stated that "[t]he effect on the plant simulator will be minimal. The simulator initial conditions will be revised to account for the increase from 3,293 to 3,339 MWt as 100% power." The licensee will also add "an annunciator point [and corresponding annunciator tile] to alert operators to Crossflow trouble." No other changes to the simulator were identified.

In the supplemental response dated May 7, 2001, the licensee clarified that simulator modifications will be made in accordance with the 1993 revision of ANSI/ANS 3.5, "Nuclear Power Plant Simulators for Use in Operator Training."

The staff finds the licensee's submittal satisfactory because the licensee has adequately described how the changes to operator actions will be addressed by the simulator and how the simulator will accommodate the changes in accordance with the requirements of ANSI/ANS Standard 3.5.

3.10.6 Human Factors - Conclusion

Based on the evaluation in Sections 3.10.1 through 3.10.5 of this safety evaluation, the staff concludes that the proposed power uprate is acceptable with respect to its impact on human factors considerations.

3.11 Accident Analysis

3.11.1 Accident Analysis - Background

Anticipated operational occurrences (AOOs) are abnormal transients which are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single failure of equipment, or a personnel error. The applicable acceptance criteria for the AOOs are based on 10 CFR Part 50, Appendix A, GDC 10, 15, and 20. GDC 10 requires, in part, that the reactor core and associated control and protection systems be designed with appropriate margin to ensure that the specified acceptable fuel design limits are not exceeded during normal operation and during AOOs. GDC 15 stipulates that these systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operating conditions and AOOs. GDC 20 specifies that a protection system be designed to initiate automatically the operation of appropriate systems to ensure that the specified fuel design limits are not exceeded during AOOs.

The Standard Review Plan (SRP) provides further guidelines: (1) pressure in the reactor coolant and main steam system should be maintained below 110% of the design values according to the ASME Code, Section III, Article NB-7000, "Overpressure Protection"; (2) fuel cladding integrity should be maintained by ensuring that the reactor core is designed to operate with appropriate margin to specified limits during normal operating conditions and AOOs; (3) an incident of moderate frequency should not generate a more serious plant condition unless other faults occur independently; and (4) an incident of moderate frequency, in combination with any single-active component failure or single operator error, should not result in the loss of function of any fission product barrier other than the fuel cladding (a limited number of fuel cladding perforations are acceptable).

3.11.2 Accident Analysis - Evaluation

Chapter 15 of the HCGS UFSAR contains the design basis analyses that evaluate the effects of an AOO resulting from changes in system parameters such as: (1) a decrease in core coolant temperature, (2) an increase in reactor pressure, (3) a decrease in reactor coolant flow rate, (4) reactivity and power distribution anomalies, (5) an increase in reactor coolant inventory, and (6) a decrease in reactor coolant inventory. The plant's responses to the most limiting transients are analyzed each reload cycle, and corresponding changes in the MCPR are added to the SLMCPR to establish the cycle-specific operating limit MCPR. A potentially limiting event is an event or an accident that has the potential to affect the core operating and safety limits.

PSEG has assessed the impact of the 1.4% power uprate on the AOOs analyzed in the HCGS UFSAR and determined that the AOOs fall into the four categories discussed below in Sections 3.11.2.1 through 3.11.2.4.

3.11.2.1 AOOs That Need To Be Reanalyzed on a Cycle-Specific Basis

The licensee stated that the following AOOs are the most limiting transients and will be reanalyzed on a cycle-specific basis using the applicable power measurement uncertainty:

- loss of feedwater heating
- feedwater controller failure-maximum demand
- generator load reject
- rod withdrawal error
- recirculation flow controller failure with increasing flow
- inadvertent HPCI startup

3.11.2.2 AOOs Bounded by the Cycle-Specific Analyses

According to PSEG, the proposed power uprate will not make a nonlimiting transient into a limiting transient; therefore, the following transients will remain bounded by the limiting cycle-specific transients.

- turbine trip
- pressure regulator failure-open
- inadvertent main steam relief valve opening

- inadvertent RHR shutdown cooling operation
- pressure regulator failure-closed
- MSIV closure
- loss of condenser vacuum
- loss of ac power
- loss of feedwater flow
- reactor recirculation pump trip
- recirculation flow controller failure-decreasing flow
- abnormal startup of idle recirculation pump

3.11.2.3 AOOs Previously Analyzed at Bounding Power Level

The licensee stated that the failure of RHR shutdown cooling transient was analyzed at power levels higher than the proposed uprate power level in spite of the reduced power measurement uncertainty. Therefore, this transient need not be reanalyzed for the uprated conditions.

3.11.2.4 AOOs Unaffected by the Proposed Power Uprate.

The licensee's submittal provided estimates of the plant operating parameters at the uprated conditions. The changes to the feed flow, steam flow, and feedwater temperature are relatively small. The licensee stated that the changes associated with the power uprate will have an insignificant impact on both the results of the transients and the Westinghouse methodology used to analyze the following transients:

- rod withdrawal error-low power
- control rod maloperation transient

3.11.3 Accident Analysis - Conclusion

The licensee concluded that the NRC-approved methodology in CENPD-300P-A remains applicable to calculate the effects of the limiting reactor transients at the uprated conditions. The staff agrees with the licensee that the plant operating parameters at the uprated condition will not change significantly. Therefore, a nonlimiting transient is not expected to become limiting or change the applicability of the NRC-approved methodology. Transients are analyzed at offrated conditions, maximum rated conditions, or higher, depending on the transient's sensitivity to core parameters such as power, core flow, and inlet subcooling. The limiting transients will be reanalyzed for the reload cycle at the off-rated or the uprated conditions, depending on the type of transient. The changes in the MCPR and the LHGR during the limiting transients will be used to establish the exposure, power, and flow dependent operating limit MCPR or LHGR. The effect of the power uprate will be accounted for and incorporated into the plant's operating limits in the cycle-specific reload analysis using the NRC-approved methodologies specified in CENPD-300-P-A. Therefore, the staff concludes that the proposed power uprate is acceptable with respect to its impact on the accident analysis.

3.12 Other Evaluations

3.12.1 Environmental Qualification (EQ)

The licensee evaluated the proposed power uprate with respect to its effect on the equipment in the HCGS EQ program for normal and accident conditions as described below.

The current normal conditions for temperature, pressure, and humidity are unchanged for the proposed power uprate operating conditions. There is no change in the high energy line break (HELB) profile. The only high-energy system with an operating pressure and temperature change is the feedwater system. The operating pressure increases from 1,185 psia to 1,188 psia, and the design operating temperature increases from 420 °F to 421 °F. The HELB profile had been generated using 1,185 psia and 425 °F as the initial condition for feedwater break analyses. Thus, the uprated feedwater temperature is bounded by the analyses and the pressure change is only 0.25%. The net change is further reduced when the pressure loss is considered at the postulated break location. Therefore, no impact on the HELB profiles is expected for the proposed 1.4% power uprate.

The licensee performed LOCA and containment response analyses assuming the reactor power was at 102% of the current rated thermal power level of 3,293 MWt. The existing analysis bounds the pressure, temperature, and humidity profiles for equipment inside the containment for the proposed power uprate operating conditions.

The licensee's application states that normal radiation inside the drywell will increase by about 2% at the proposed 1.4% uprated power level and that accident levels of radiation inside containment are unchanged by the power uprate. The licensee's application also states that the qualified radiation value envelopes the normal radiation dose (including the 2% increase) plus the postulated accident dose with margin as specified in IEEE-323-1974.

Based on our review of the licensee's application, as set forth above, the staff concludes that the proposed power uprate does not have any affect on the equipment in the HCGS EQ program.

3.12.2 Station Blackout (SBO)

The licensee reviewed the SBO plant response and coping evaluations that were originally performed to satisfy the requirements in 10 CFR 50.63. The coping duration of 4 hours is still valid at the 1.4% power uprate without any changes to the systems and equipment used to respond to an SBO event.

A slight increase in decay heat may have a small impact on suppression pool and drywell temperatures (about 1 °F), but this increase is insignificant. The licensee analyzed the impact of this increase on the condensate water volume necessary for coping and the temperature heat-up in the areas which contain equipment necessary to mitigate the SBO event, and concluded that no changes to the systems and equipment used to cope with an SBO event are required. Computer simulations were used to demonstrate that the minimum maintained condensate storage tank inventory remains adequate to provide makeup to the RPV for 4 hours during an SBO event at the 1.4% power uprate. Temperature in the HPCI and RCIC rooms and other areas containing SBO coping equipment is not affected by power uprate.

Based on our review of the licensee's application, as set forth above, the staff finds that the impact on the licensee's ability for coping with an SBO event due to plant operations at the proposed 1.4% uprated power level will be insignificant.

3.12.3 HELB

The licensee stated that the proposed 1.4% power increase will result in a slight increase in the feedwater system pressure (from 1,185 psia to 1,188 psia) and temperature (420 °F to 421 °F). The original feedwater line break analyses used 1,185 psia and 425 °F. Thus, the uprated feedwater temperature is bounded by the original analyses, and the slight increase in pressure (0.25%) is insignificant for the mass and energy release rates resulting from a feedwater break outside of containment.

Based on our review of the licensee's application, as set forth above, the staff concludes that the existing analyses for HELB will remain bounding and will be acceptable for plant operations at the proposed uprated power level.

3.12.4 ATWS

An ATWS is an anticipated operational occurrence followed by failure of the reactor trip portion of the protection system. The requirements for reduction of risk from ATWS events are specified in 10 CFR Part 62. The regulation requires BWR facilities to have the following mitigating features for an ATWS event:

(1) a SLC system with the capability of injecting into the reactor pressure vessel a borated water solution such that the resulting reactivity control is at least equivalent to the control obtained by injecting 86 gpm of a 13-weight-percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside-diameter reactor vessel,

(2) an alternate rod injection (ARI) system that is designed to perform its function in a reliable manner and that is independent (from the existing reactor trip system) from sensor output to the final actuation device, and

(3) equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

BWR facilities meet the ATWS acceptance criteria to demonstrate the ability to withstand an ATWS event, that is, maintaining the fuel integrity (the core and fuel must maintain a coolable geometry), the primary system integrity (the peak reactor vessel pressure remains below 1,500 psig), and the containment integrity (the containment pressure must not exceed the design limit).

The SLC system provides the means of attaining and maintaining a cold shutdown condition, assuming no control rod movement, as required by GDC 26. The ATWS analyses assume the SLC will inject at a specified time to bring the reactor to and maintain it at cold shutdown during an AOO. For every reload, the licensee must evaluate how plant modifications, reload core designs, changes in fuel design, and other reactor operating changes affect the applicability of the ATWS analysis of record.

PSEG stated that HCGS meets the ATWS requirements, having an automatic recirculation pump trip (ATWS-RPT) to produce negative reactivity and an ARI and SLC system to shut down the reactor. According to the licensee, the calculated maximum vessel bottom pressure for the MSIV event is 1,425 psig, which is below the ATWS emergency condition allowable pressure of 1,500 psig. The licensee added that the ATWS analysis conservatively assumed initial operating conditions of 100% power and 87% core flow because the effectiveness of the ATWS-RPT is less pronounced at maximum power and higher flow conditions.

The ATWS analysis also assumed a single upper limit SRV setpoint of 1,250 psig for the 13 available SRVs with 1 SRV out of service. The actual SRV setpoints are 1,108 psig for four SRVs, 1,120 psig for five SRVs, and 1,130 psig for five SRVs. The licensee determined that these conservatisms would more than offset the effect of the 1.4% power uprate and that the proposed power uprate will have no effect on the ability of the SRV's to mitigate the consequences of an ATWS.

The limiting ATWS analysis of record is based on a GE fuel core. The power uprate reference core consists of ABB SVEA-96+ fuel and resident GE9B fuel. Future cycles are expected to use exclusively SVEA-96+ fuel. Therefore, the licensee considered the implications of exclusively using SVEA-96+ for the limiting ATWS analysis of record. The licensee stated that the SVEA-96+ fuel has a lower LHGR than the larger diameter rods of the resident GE fuel. Consequently, the heatup characteristics of a mixed core or full SVEA-96+ core during an ATWS event will be less limiting than a GE9B core. The licensee referred to the Cycle 10 ATWS evaluation, which showed that the SVEA-96+ fuel design and the Cycle 10 mixed core both met the ATWS criteria. Accordingly, the licensee stated that the heatup characteristics for a mixed core were less limiting than heatup characteristics of the GE 8x8 core used in the plant licensing basis ATWS analysis. This conclusion applies to all future cycles since additional margin will be gained as the resident fuel is discharged and the core becomes fully loaded with SVEA-96+. The licensee also considered whether the introduction of the SVEA-96+ fuel for the power uprate would affect the ATWS evaluation and determined that the SVEA-96+ evaluation would remain applicable up to a power uprate of 1.5%, which bounds the proposed 1.4% power uprate.

Section 9.5 of the NRC-approved topical report CENPD-300 states that the ATWS analysis is not performed for every reload. ATWS is reanalyzed only if plant modifications have the potential to challenge the event acceptance limits. The report stated that each ABB fuel design introduced into a core will be confirmed to have heatup characteristics that are less limiting than those assumed in the plant licensing basis ATWS analysis. Appendix D of the report states that the smaller rods in the SVEA-96+ fuel design relative to the GE 8x8 or 9x9 fuel designs result in lower LHGR and core heat flux characteristics. The appendix compares the MSIV closure pressure and heat response against the response of resident fuel. The topical report concludes that reload cores of SVEA-96+ fuel are not expected to adversely change the core average response during an ATWS event. Although the approving safety evaluation did not specifically review Appendix D of the report, it did approve the methodology used to evaluate whether an ATWS analysis based on another vendor's resident core would be applicable to a mixed core or an exclusively SVEA-96+ reload core. Since the licensee's evaluation is based on an NRC-approved methodology and confirms the sensitivity studies in Appendix D of the topical report, the staff accepts that the HCGS limiting MSIV closure/ATWS analysis of record can be based on a GE core.

The proposed 1.4% power uprate will produce slightly higher peak pressures, heatup response, and decay heat during an ATWS. Moreover, the licensee confirmed that the SLC system will be able to inject the required flow rate for the limiting MSIV closure ATWS analysis because the peak pressure will dissipate before the SLC system is assumed to inject. For this event, the SLC system discharge pressure will not reach 1400 psig. However, the licensee did not perform plant-specific ATWS analyses for all of the other AOO events that were generically analyzed in NEDE-24222, Volume 2. General Electric (GE) performed generic ATWS evaluations for MSIV closure, pressure regulator failure-open (PRFO), loss of offsite power (LOOP), and inadvertent opening of a relief valve (IORV) for power uprates up to 5% above the original licensed power level. While the NRC-approved CENPD-300-(P)(A) report established a methodology to evaluate the response of the SVEA-96+ fuel core with GE resident fuel for the MSIV closure event, similar evaluations were not performed for the other ATWS scenarios. Therefore, the applicability of GE's generic ATWS analysis for the PRFO, LOOP or IORV events had not been considered by the fuel vendor and the licensee. The staff, therefore, cannot disposition all of the ATWS scenarios for HCGS based on GE's generic ATWS analyses in determining whether the SLC system will be able to inject for all of the limiting ATWS AOOs without exceeding the SLC relief valve setting of 1400 psig.

However, HCGS has not implemented any power uprate before this one, so the dome pressure and the SRV setpoints have not changed. Therefore, considering the conservatism in the current HCGS ATWS analysis (all SRVs lift at 1,250 psig), the available peak pressure margin, the lower LHGR of the SVEA-96+ fuel, and the small operating parameter changes associated with the power uprate, the staff accepts the licensee's conclusions. If HCGS decides to pursue a larger power uprate at a future date, the staff expects that the licensee will perform a complete set of ATWS analyses to determine the limiting scenarios and establish that the plant can meet all of the 10 CFR 50.62 requirements.

3.12.5 Safety-Related Valves

The licensee reviewed its motor-operated-valve (MOV) program and stated that the MOV evaluation at HCGS was performed using the worst-case parameters from the accident analyses and, therefore, bounds the proposed 1.4% power uprate condition. The licensee evaluated its commitments relating to Generic Letter (GL) 95-07, associated with the pressure locking and thermal binding of safety-related power-operated gate valves that are required to open to perform their intended safety function. The licensee found that the existing analysis conditions remain bounding for the 1.4% power uprate. The licensee also evaluated its response relating to the GL 96-06 program regarding the overpressurization of isolated piping segments. The licensee concluded that the existing evaluation for GL 96-06 was performed at 102% of current rated power and is, therefore, bounding for the 1.4% proposed power uprate. On the basis of the above review, the staff concurs with the licensee's conclusions that the power uprate will have no adverse effects on safety-related valves and that the licensee's conclusions from the GL 95-07, GL 96-06, and GL 89-10 programs remain valid.

3.12.6 Other Evaluations - Conclusion

Based on the evaluation in Sections 3.12.1 through 3.12.5 of this safety evaluation, the staff concludes that the proposed power uprate is acceptable with respect to its impact on HCGS analyses and programs pertaining to EQ, SBO, HELB, ATWS, and safety-related valves.

3.13 FOL and TS Changes

3.13.1 Changes To Reflect Increase in Licensed Core Power Level

The licensee proposed to revise the FOL and TSs as follows to reflect the increase in licensed power level from 3,293 MWt to 3,339 MWt:

(1) Paragraph 2.C.(1) in Facility Operating License NPF-57, “Maximum Power Level,” would be revised to authorize operation of the facility at reactor core power levels not in excess of 3,339 MWt (100% of rated power).

(2) The definition of RATED THERMAL POWER in TS 1.35 would be revised to state that the rated thermal power shall be a total reactor core heat transfer rate to the reactor coolant of 3,339 MWt.

(3) TS 6.9.1.9, “Core Operating Limits Report,” would be revised to add a reference to Topical Report CENPD-397-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology."

The FOL and TS changes reflect the proposed increase in licensed power level based on installation of the Crossflow UFM system. Based on the evaluations discussed in Sections 3.1 through 3.12 of this safety evaluation, the staff concludes that the above-described changes to the FOL and TSs are acceptable.

3.13.2 Changes to Pressure-Temperature (P-T) Limits

The licensee’s submittal dated December 1, 2000, proposed to revise the following TSs to address RPV integrity issues associated with the proposed power uprate:

(1) TS Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 (P-T limit curves for hydrostatic testing, non-nuclear heatup and cooldown, and core critical heatup and cooldown) would be revised to support the increase in core power level based on uprated fluence projections.

(2) Surveillance Requirement 4.4.6.1.4 would be revised to be made consistent with the limit on reactor vessel flange and head flange temperature in TS 3.4.6.1.d.

As discussed below, the NRC staff reviewed the proposed changes to the P-T limit curves with respect to the neutron fluence values that were used and RPV fracture toughness.

Neutron Fluence

The proposed P-T limit curves in the submittal dated December 1, 2000, were developed with an applicability of 32 effective full-power years (EFPY) of operation, which corresponds to the end of the current license. In a teleconference between the NRC staff and the licensee on February 1, 2001, the staff indicated that there were unresolved technical issues regarding the methodology used to derive the fluence values for the proposed amendment. This methodology is the subject of GE Topical Report NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," which is currently under review by the NRC staff. The staff concluded that the technical issues must be resolved in order to justify applying the fluence values for 32 EFPY. As an interim solution, in a submittal dated February 12, 2001, the licensee added a note to the proposed P-T limit curves stating that the curves are only applicable to the end of Cycle 11 (i.e., the next operating cycle).

The fluence values used for the P-T curves are for 32 EFPY, while the estimated vessel fluence to the end of cycle 11 is 42% of the value for 32 EFPY. The NRC staff finds that the fluence values used by the licensee are conservative for Cycle 11 operation. Therefore, the staff concludes that there is a reasonable assurance of safety and the proposed fluence values are acceptable for use in developing the P-T limit curves for Cycle 11 operation.

Fracture Toughness

The licensee's submittal dated December 1, 2000, provided the following information regarding their evaluation of the proposed P-T limit curve changes with respect to RPV fracture toughness:

- The curves were developed using the methodology specified in ASME Code Cases N-588 and N-640, as well as the 1989 ASME Code, ASME Section XI, Appendix G, and 10 CFR Part 50, Appendix G. The improvement realized from the Code Case methodology is as much as 60°F, and is primarily obtained from using the critical fracture toughness, K_{IC}, in accordance with Code Case N-640. Pressure and temperature instrument uncertainties are included in the revised curves.

- Adjusted reference temperature at the nil ductility transition (ART_{NDT}) values were developed for the RPV materials in accordance with Regulatory Guide 1.99, Revision 2, based on projected fluence values which were increased in proportion to the increase in rated power. The calculated increase was based on the conservative assumption that the power uprate was initiated at the beginning of the current fuel cycle. The ART_{NDT} is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term by application of Regulatory Guide 1.99.

- Three regions of the RPV were evaluated to develop the revised P-T limit curves: (1) the beltline region, (2) the bottom head region, and (3) the feedwater nozzle/upper vessel region. These regions bound all other regions with respect to brittle fracture.

- Upper shelf energy (USE) calculations were performed and it was confirmed that all USE values are greater than 50 ft-lb throughout RPV life, as required by 10 CFR Part 50, Appendix G.

- Surveillance Requirement (SR) 4.4.6.1.4 is being revised to be made consistent with the TS limit on reactor vessel flange and head flange metal temperature. TS 3.4.6.1.d requires that reactor vessel flange and head flange metal temperatures be maintained greater than or equal to 79 °F when reactor vessel head bolting studs are under tension. SR 4.4.6.1.4.a currently requires flange temperatures to be verified to be greater than 70 °F in Operational Condition 4 once per 12 hours when RCS temperature is less than or equal to 100 °F and once per 30 minutes when RCS temperature is 80 °F. SR 4.4.6.1.4 is being revised to refer to the TS limit for minimum flange temperature, instead of 70 °F. The limits on RCS temperature are also being revised to assure margins to temperature-pressure limits are maintained.

As discussed in Section 3.2.4 of this safety evaluation, the licensee's use of ASME Code Cases N-588 and N-640 was previously approved by an exemption to NRC regulations dated July 12, 2001.

The NRC staff has evaluated the licensee's submittals to determine whether the proposed licensing action would reduce the margins of safety that have been established in the licensing basis to ensure the structural integrity of the HCGS reactor coolant pressure boundary and, in particular, to ensure the integrity of the RPV. For the proposed 1.4% power uprate, RPV normal operating parameters (pressure and temperature) remain unchanged from the current operating condition. The planned approach to achieve 1.4% increase in rated power requires no increase in maximum core flow. Current design assessments show significant design margins in reactor integrity analyses. These margins are not affected by the proposed power uprate because the loading conditions are either unchanged or are bounded by the analyzed loading conditions. Based on the above, the NRC staff finds the proposed TS changes, as described above, will continue to provide an acceptable margin of safety and, therefore, are acceptable. As discussed previously in the evaluation of the neutron fluence values, the P-T limit curves are only applicable until the end of Cycle 11 operation.

3.13.3 Editorial Changes

The licensee proposed to make the following editorial change to the TSs:

(1) The TS Index is being revised to correctly show the page numbers for Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3.

The staff considers these changes to be administrative in nature and acceptable.

3.13.4 TS Bases Changes

The licensee proposed to make the following changes to the TS Bases:

(1) TS Bases Table B 3/4.4.6-1 and TS Bases Figure B 3/4 4.6-1 would be revised consistent with the fluence value changes associated with development of the revised P-T limit curves.

(2) TS Bases 3/4.4.6 would be revised to change the reference to ASME Code Section III, Appendix G, to ASME Code Section XI, Appendix G, and to add references to ASME Code Cases N-588 and N-640.

The staff notes that the proposed TS Bases changes are consistent with the proposed license amendment.

3.14 Evaluation Summary

Based on the evaluation in Sections 3.1 through 3.13 of this safety evaluation, the staff concludes that the proposed amendment is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the *Federal Register* on June 22, 2001 (66 FR 33583). Accordingly, based upon the Environmental Assessment, the staff has determined that issuance of the amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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